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7. Abstract

Radionuclides in the Hanford tank waste are ranked relative to their consequences for the Low-Level Tank Waste program. The ranking identifies key radionuclides where further study is merited. In addition to potential consequences for intrude and drinking-water scenarios supporting low-level waste activities, a ranking based on shielding criteria is provided. The radionuclide production inventories are based on a new and independent ORIGEN2 calculation representing the operation of all Hanford single-pass reactors and the N Reactor.

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Consequence Ranking of Radionuclides in Hanford Tank Waste

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1.0 INTRODUCTION

Sampling and analysis of Hanford Tank Waste is very expensive, and a reliable assessment of the most important contaminants is paramount. Although many detailed studies have been completed in the past, a new examination was deemed worthwhile. Part of the difficulty with past studies is the sheer volume of information. These studies are based on numerous overlapping assumptions and data, and it is sometimes difficult to assess the overall quality of these rankings. As the Hanford cleanup effort becomes a reality, it is timely to review which contaminants are of most concern.

The strategy of this study is to provide a ranking basis that includes the entire waste stream process beginning with calculations of the production reactor inventories through final disposal of the waste. To maintain a reasonable scope, the focus is on radionuclides especially in the context of low-level waste performance. As much as possible, generally valid assumptions are made in preference to more precise and detailed evaluations. The merit in this approach is that the results derive from clear and concise bases, assumptions, and data.

The dose consequences for the low-level waste follow from a natural sequence of operations that are documented in the following sections. First, the total radionuclide inventories are calculated based on the complete operating history of all the Hanford production reactors including N Reactor. Next a series of "splits" or reduction factors are identified that describe, at each step, the fraction of each radionuclide that contributes to the low-level waste stream. For example, plutonium and uranium were extracted from the waste sent to the double and single-shelled tanks as well as some other isotopes including cesium and strontium. Pretreatment of the waste will then further reduce the radionuclides in the low-level waste stream, notably cesium, followed by vitrification that is expected to reduce (i.e., volatilize) iodine and carbon 14. Chemical retardation in the soil gives the final inventory reduction prior to entering the unconfined aquifer. The last step in the analysis is the conversion of contaminant concentrations to dose consequences.

2.0 RADIONUCLIDE PRODUCTION CALCULATIONS

A new calculation of the total Hanford radionuclide production was completed using the ORIGEN2 code (Croff 1980). The calculation is documented by the input runstream given in Appendix A. To understand the consequences of simplifying assumptions that were necessarily made, a brief discussion of the radionuclide production process is appropriate.

The overall integrated power of the Hanford production reactors is well known (Roblyer 1994). Since the energy released per fission is also well known, the total number of fissions is thus established. The total production of most of the important fission-product radionuclides is then readily obtained from their fission-product yields, independent of other assumptions.

The production of transuranic nuclides is more complicated; although the relatively low burnup of the production reactors simplifies this problem as well. The main pathway to most of the transuranic radionuclides is through the production of ^{239}Pu via transmutation of ^{238}U. Thus, the well-documented Pu conversion factor provides a good check on the first step in the production of most transuranics. A significant exception to this is ^{237}Np which arises from a two-step transmutation of ^{235}U and via (n, 2n) from ^{238}U.

Unlike the other categories, the calculation of activation products is fraught with difficulties. The amount of trace impurities may be poorly known or documented, and the activation rate is difficult to calculate. Thus, the production calculation of radionuclides such as ^{14}C and ^{60}Co is highly uncertain without detailed studies.

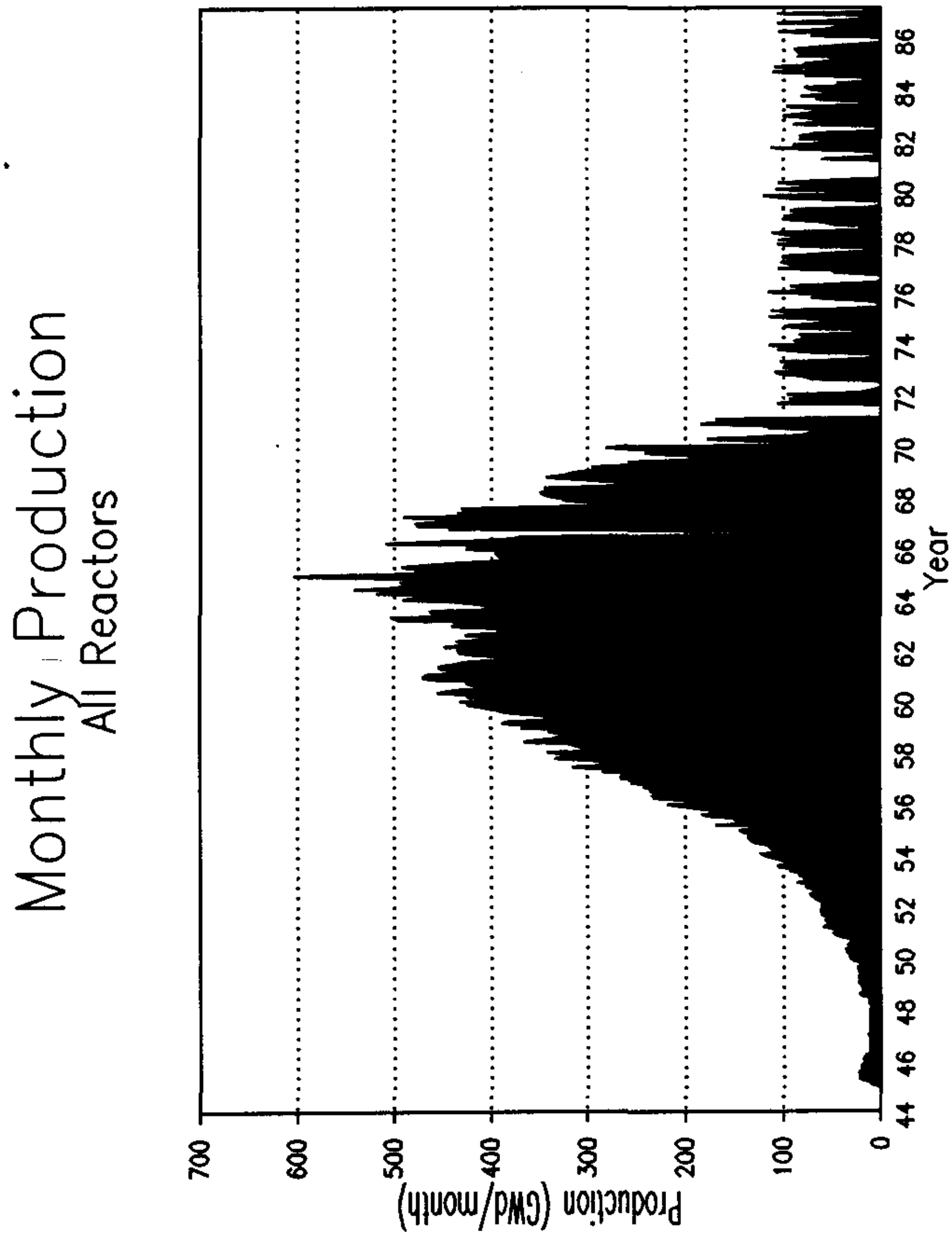
The explicit ORIGEN2 production model consists of three sections. The first section models the single-pass reactors, the second section models the N Reactor history, and the final section combines the results into composite values decayed to 1990. Additional decay steps are added to obtain production inventories at 2010 and 2030 (approximately representing treatment operations and closure respectively), then +100 y and +500 y (for intruder scenarios), and finally +5000 y, +10,000 y and +100,000 y (for long term dose consequences). Not all times are reported here.

The complete power history for both the single-pass reactors and N Reactor is shown in Figure 1. The integrated operating powers add up to a total of 67.1×10^6 MWd for the single-pass reactors and about 14.6×10^6 MWd for N Reactor (the latter value includes operations through the time of N Reactor shutdown in 1987). These values give a strong constraint on total production of fission-products such as ^{99}Tc. The specific power for both the single-pass model and N Reactor was fixed at 10 MW/MTU, a value typical of N Reactor values.

2.1 SINGLE-PASS REACTORS

The ORIGEN2 model of the single-pass reactors represents the power history as a power histogram of 4 y intervals. The fuel is assumed to be natural uranium. Although some enriched spiked fuel was used, this level of detail is not warranted here. One metric ton of natural uranium is first irradiated with a specific power of 10 MW/MTU for the relatively low burnup of 800 MWd/MTU. It is then decayed for 100 d and reprocessed with extraction coefficients of 99% for plutonium and 99.5% for uranium (only plutonium and uranium had non-zero coefficients). The complete production histogram was obtained by repeatedly rescaling this unit and adding it to the accumulated results. At each step prior to the final one, the accumulated results were

Figure 1. Power History for Both Single-Pass and N Reactors.



decayed 4 years to take them to the next time step. The final results are then renormalized to the 67.1 GWd total integrated power. The ORIGEN2 cross sections used were developed for N-Reactor Mark IV fuel, and in order to match the plutonium conversion ratios, the ^{235}U fission cross section was adjusted to 100.0 barns. The rational for this adjustment is as follows.

The burnup, B (MWd/MTU), is given by

$$B = St$$

for a constant specific power, S (MW/MTU). The specific power is given by

$$S = E_f(\sigma_f^{235}\phi_o n_o^{235})$$

where E_f (MeV) is the energy released per fission, and the factors within parentheses represent the fission rate given in terms of the initial neutron flux ϕ_o, the initial amount n_o^{235} of ^{235}U, and its fission cross section σ_f^{235}.

The ^{239}Pu production is given in terms of the initial ^{238}U inventory by

$$n^{239}(t) = n_o^{238}\left(\frac{\phi_o\sigma_c^{238}}{\phi_o\sigma_a^{239}}\right)\left(1-e^{-\sigma_a^{239}\phi_o t}\right)$$

where the notation is the same as above except for the subscripts c and a to denote capture and absorption cross sections, respectively.

Keeping only the first order term in the exponential and combining the result with the previous equations readily yields the following expression for the plutonium conversion factor:

$$\frac{n^{239}}{B} = \frac{1}{E_f}\left(\frac{\sigma_c^{238}}{\sigma_f^{235}}\right)\left(\frac{1}{\epsilon}-1\right)$$

where the enrichment ϵ is defined by

$$\epsilon = \frac{n^{235}}{n^{235}+n^{238}}$$

Since the neglected higher order terms represent, in part, the transmutation of ^{239}Pu to other plutonium isotopes, this approximation is consistent with reinterpreting the ^{239}Pu conversion factor as the total plutonium conversion factor.

As this simple expression for the plutonium conversion factor shows, for a fixed enrichment only the ^{238}U capture and the ^{235}U fission cross sections are important. (The energy-released per fission E_f is fixed near 200 MeV.)

In addition to the assumed natural uranium fuel, the cladding was represented by 5.0 wt% aluminum. Cladding impurities were not included; however, fuel impurities developed for N Reactor were arbitrarily added (Hedengren and Goldberg 1987). As noted, the calculation of trace impurities can be difficult, and this question may need to be revisited.

2.2 N REACTOR

The ORIGEN2 N Reactor simulation was patterned after the single-pass reactor model in using five-year intervals to model the power history. The same uranium and plutonium extraction efficiencies were used; however, the decay time from discharge to reprocessing was increased from 100 days to 300 days. This adjustment is significant because it increases the amount of ^{241}Am that builds up from the decay of ^{241}Pu. The fuel was assumed to be enriched to 0.95% and included ^{234}U and ^{236}U. (Much of the N Reactor fuel was enriched to 0.94% with spiked values to 1.25%.). Fuel impurities and cladding were included based on earlier N Reactor calculations (Hedengren and Goldberg 1980). The cladding model represented 7.0 wt% Zircaloy and also included trace impurities.

The ^{235}U fission cross section was reduced to 55 barns which is a little lower than typical N Reactor values. As for the single-pass model, this value was used to adjust the plutonium conversion rate. An alternative approach would have been to modify the enrichment slightly from the assumed value of 0.95%.

2.3 COMPOSITE RESULTS

Plutonium production values and uranium inventories provide good global checks on the ORIGEN2 calculational model. The values calculated with ORIGEN2 are given in Table 1. (The burnup values in column 4 are input values.) Note that the uranium values are not independent input but are derived as a consequence of normalizing to the given total burnups. The uranium values also provide an indirect check on the assumed specific powers.

The plutonium production numbers are in excellent agreement with recently reported values (Roblyer 1994). The single-pass ORIGEN2 value is about 2% lower while the N Reactor value is about 10% higher giving total production values well within 1%. (Note that most of the plutonium production is from the single-pass reactors.) The total uranium production given in Table 1 is 6% below fuel discharge values obtained from a related database reported as 96.4×10^3 MT). This small difference could be easily corrected by adjusting the specific power.

Table 1. Global Power and Production Inventories Calculated from ORIGEN2. Burnup values in column 4 are input parameters.

Reactor	U (g)	Pu (g)	BU (MWd)	Pu Conv. (g/MWd)
Single-pass	8.37×10^{10}	55.0×10^{6}	67.1×10^{6}	0.820
N Reactor	0.73×10^{10}	12.1×10^{6}	14.6×10^{6}	0.829
Total	9.10×10^{10}	67.1×10^{6}	81.7×10^{6}	

2.4 COMPARISONS WITH OTHER STUDIES

The comparisons just discussed provide a good global check on the ORIGEN2 production model. More detailed comparisons with prior studies are presented next. However, the comparison values are not as well supported. The assumptions are weaker or not well documented; indeed it is this very problem that, in part, motivated developing a new global production model.

The comparison chosen is based on the Hanford Defense Waste Environmental Impact Statement, [HDW-EIS] (DOE 1987) and a supporting document, *Hanford Defense Waste Disposal Alternatives* (RHO 1985). This report focused on disposal alternatives for the six waste types identified at the Hanford Site (existing tank waste, transuranic-contaminated soil sites, pre-1970 TRU buried solid waste sites, retrievably stored and newly generated solid TRU waste, strontium and cesium capsules, and future tank waste).

Table 2 shows the predicted Site waste inventory (in Curies) for each radionuclide modeled by ORIGEN2 and included in the *Hanford Defense Waste Disposal Alternatives*. Since the ORIGEN2 estimates were decayed to 1990, each of the inventory estimates in (RHO 1985) were decay-adjusted to the end of 1990 to facilitate the comparison. This allows direct comparison between each set of estimates. To further facilitate the comparison, ORIGEN2 values were obtained for production through 1971 while the future tank waste category from (RHO 1985) was not included.

A summary of Table 2 follows. The first column lists each nuclide that can be compared.

The second column contains the ORIGEN2 values with the following assumptions: All single-pass reactor operations and N Reactor production through 1971 with 99.5% uranium and 99.0% plutonium extraction coefficients.

The third column displays the reported inventory in the tanks based on TRAC data as referenced in (RHO 1985).

The fourth column gives the total inventory of the four other waste types (transuranic-contaminated soil sites, pre-1970 TRU buried solid waste sites, retrievably stored and newly generated solid TRU waste, strontium and cesium capsules), with each type decayed-corrected to the end of 1990.

The "Total" (5th column) is the sum of the existing tank waste (column 3) and the "other four types" (column 4). The remaining waste type (future tank waste) was not included in the total because the values for this waste type assumed that N Reactor and the Plutonium Finishing Plant operated through 1995, and PUREX continued operation beyond 1995.

The comparison is shown in the last column of Table 2 as the ratio of the "Total" values from RHO (1985) to the ORIGEN2 results.

The following observations are made (Ratios to the ORIGEN2 values are shown in parentheses).

The ^{99}Tc values are in close agreement (0.94). Another radionuclide ^{129}I with potentially high environmental consequences is in fairly good agreement (0.75) but not as close as expected. For ^{137}Cs and ^{90}Sr the agreement is reasonable (0.71, 0.88) but again not as good as expected. The ^{137}Cs to ^{90}Sr ratio indicates that the accounting in the RHO (1985) values is suspect. The RHO (1985) ratio is 0.92 (8.36×10^7 Ci / 9.09×10^7 Ci) versus the ORIGEN2 ratio of 1.16 (1.19×10^8 Ci / 1.03×10^8 Ci).

The plutonium (^{239}Pu, ^{240}Pu) and uranium (^{235}U, ^{238}U) estimates in (RHO 1985) are approximately a factor of three (2.47 to 3.41) higher than the estimates generated by ORIGEN2. Part of the uranium difference may be attributed to the relatively high (99.5%) extraction efficiency assumed. This discrepancy is not unexpected. The results are sensitive to the extraction efficiencies, and reliable uranium and plutonium values have not been publicly available.

Many of the other radionuclides are more difficult to calculate, and the agreement is inconsistent. For example, another environmentally sensitive radionuclide, ^{14}C, is off by a factor of seven. However, ^{14}C is an activation product whose production is known to be very sensitive to trace impurities such as ^{14}N. Although previous studies have investigated this issue, no attempt was made to adjust the ORIGEN2 calculation in this screening study. The RHO 1985 value is a better estimate.

The ^{237}Np value from ORIGEN2 is higher by a factor of six. This isotope is sensitive to the ORIGEN2 cross section data which may be incorrect. Furthermore, the recovery of neptunium during reprocessing was not included in the ORIGEN2 model. Again, the RHO 1985 value is much more likely to be correct.

Another discrepancy of concern is the result for ^{233}U. The severe underestimate by ORIGEN2 is most likely a consequence of not including known thoria campaigns in the ORIGEN2 model. This process should be included in revised models. A newer model could also include the use of recycled uranium which would effect the ^{234}U and ^{236}U values as well as ^{237}Np.

Table 2. Comparison of RHO (1985) values to ORIGEN2. The RHO (1985) report is the supporting document for the Hanford Defense Waste Environmental Impact Statement (HDW-EIS). The ORIGEN2 values represent production only through 1971 to effect a more direct comparison. All values are given in curies decayed to the end of 1990.

Radio-nuclide	ORIGEN2	RHO 85 (In Tank)	RHO 85 (other)	Total (In tank + other)	Ratio Total/ORIGEN2
Am-241	3.42E+04	4.40E+04	1.36E+04	5.76E+04	1.68
Am-243	1.92E+01	3.40E+01	4.50E-02	3.40E+01	1.78
C-14	6.89E+02	5.00E+03	4.80E+00	5.00E+03	7.27
Cm-244	1.68E+03	1.70E+02	0.00E+00	1.70E+02	0.10
Cs-135	1.02E+03	1.40E+02	0.00E+00	1.40E+02	0.14
Cs-137	1.19E+08	2.40E+07	5.96E+07	8.36E+07	0.71
I-129	6.15E+01	4.60E+01	0.00E+00	4.60E+01	0.75
Ni-63	1.67E+04	3.20E+05	0.00E+00	3.20E+05	19.16
Np-237	4.10E+02	6.50E+01	8.00E-02	6.51E+01	0.16
Pu-238	2.37E+03	4.70E+02	3.30E+04	3.34E+04	14.10
Pu-239	3.49E+04	2.20E+04	7.90E+04	1.01E+05	2.89
Pu-240	6.24E+03	5.30E+03	1.01E+04	1.54E+04	2.47
Pu-241	8.84E+04	5.80E+04	2.69E+05	3.27E+05	3.70
Ra-226	1.91E-03	3.30E-07	0.00E+00	3.30E-07	<0.01
Sm-151	3.49E+06	8.50E+05	0.00E+00	8.50E+05	0.24
Sn-126	1.47E+03	7.60E+02	0.00E+00	7.60E+02	0.52
Sr-90	1.03E+08	5.60E+07	3.49E+07	9.09E+07	0.88
Tc-99	3.19E+04	3.00E+04	0.00E+00	3.00E+04	0.94
Th-230	1.73E-01	5.90E-05	0.00E+00	5.90E-05	<0.01
U-233	5.09E-02	8.30E-03	6.40E+00	6.41E+00	125.90
U-234	1.42E+02	1.70E-01	5.80E+00	5.97E+00	0.04
U-235	5.92E+00	2.00E+01	1.60E-01	2.02E+01	3.41
U-238	1.44E+02	4.80E+02	4.30E+00	4.84E+02	3.36
Zr-93	9.84E+00	4.40E+03	0.00E+00	4.40E+03	447.1

3.0 WASTE REDUCTION FACTORS

Starting from the ORIGEN2 production values, reduction factors in the radionuclide inventories are defined for three phases in the low-level waste stream: reductions from the production inventories to the tank values, reductions from the tank inventories to the low-level glass, and finally reductions from retardation in the soil. These reduction factors are denoted by r_T (to tank), r_G (to glass), and r_W (to well), respectively. Reductions from radioactive decay are accounted for in ORIGEN2. As noted, reductions in uranium and plutonium via reprocessing are explicitly treated in ORIGEN2 rather than as separate reduction factors.

Both cesium and strontium were partially encapsulated. Based on the HDW-EIS, the tank inventories for these two elements were reduced by the factors 0.56 and 0.77, respectively. The corresponding daughter elements, barium and yttrium, were scaled by the same factors (see Table 3).

The strategy in this ranking assessment is to focus attention on those contaminants where additional study is required. Consistent with this effort, is the application of reduction factors only when they are reasonably well known (or at least conservative). Thus, for a given contaminant, a high ranking may reflect either a high potential consequence or simply a lack of specific data at this point. Several contaminants are in this latter category. The blank entries in Table 3 reflect a lack of information (reduction factor = 1).

The radionuclides ^{14}C, ^{99}Tc, and ^{129}I all have potential losses (reduction factors) not accounted here. They may be volatile under some conditions and be lost, for example, in evaporators and boiling tanks. Furthermore, technetium has a known affinity for uranium, and significant fractions may have been sent offsite with uranium or entered the soil. Tank leaks as well as intentional distributions to cribs and other facilities represent other depositions not credited here. Some of these are noted in the "other" category in Table 2.

Table 3. Waste Reduction Factors.

Element	Prior to Tank[a,b]	Pretreatment[b,c]	k_d(ml/g)[d]
C		0.01	
Cs	0.56	0.01	26.0
Ba	0.56	0.01	26.0
Sr	0.77	0.04	1.0
Y	0.77	0.04	1.0
Tc		0.60	
I		0.10	
U		0.06	
Np		0.10	1.0
Pu		0.05	21.0
Am		0.06	6.0
Sn			99.0
Th			99.0
Pb			99.0

[a](DOE 1987)
[b]Blanks denote an assumed factor of 1.
[c](Boldt 1994)
[d]Distribution coefficient used to calculate a reduction factor. (Blanks denote an assumed value of 0.)

3.1 EXPECTATIONS FROM THE PRETREATMENT

The tank wastes will undergo pretreatment to separate the wastes into low-level and high-level components prior to vitrification. This includes a sludge wash and other processing steps. Current estimates for these reductions in the low-level waste stream are taken from (Boldt 1994) and are also included in Table 3. The largest reduction factors are for carbon, cesium, strontium, uranium, plutonium, and americium. Large reductions are also expected for neptunium and iodine but not currently technetium.

3.2 EXPECTATIONS FROM CONTAMINANT IN THE SOIL

The transport of emplaced waste to a potential drinking water site is a significant contributor to estimated health consequences. The drinking water dose was thus selected as one basis for ranking the important radionuclides. Although there are multiple mechanisms that may impede contaminant transport, chemical retardation in the soil can be very significant and is used here to represent a waste reduction factor in going from the emplaced waste to the concentration in drinking water.

Simple formulas for estimating this effect are given in Appendix B. Distribution coefficients needed to quantify it are given in Table 3. The values in Table 3 do not represent either best estimates or conservative bounds. They are values typical of other studies. Results in the next section are given both with and without the effect of retardation so that one can explicitly note the effect. Three of the elements in Table 3, tin, thorium, and lead, were arbitrarily assigned a distribution coefficient k_d of 99.0. Thorium and lead are acknowledged to be highly retarded but specific values were not readily available. The value for tin reflects the strong expectation that it will precipitate in the Hanford soil chemical environment.

4.0 CONSEQUENCE RANKINGS

Three categories of consequences are examined, drinking water dose, intruder dose, and shielding dose rates. The rankings are all relative. Absolute activities in curies are given in the Appendix C for reference, decayed to the end of 1990. The projected waste includes all of N Reactor operations. Although a significant fraction of the N Reactor spent fuel will not be reprocessed, the rankings given here should not be generally altered. Some changes are possible as for ^{241}Am where a significant portion resides in the unprocessed N Reactor fuel.

4.1 DRINKING WATER

For each radionuclide, the drinking water consequence, denoted by Q_W, is determined by multiplying the activity A (computed by ORIGEN2) by the waste tank reduction factor, r_T, the pretreatment reduction factor, r_G, the reduction factor, r_W, representing retardation in the soil, and finally by the specific drinking water dose conversion factor, f_D:

$$Q_W = f_D(r_T r_G r_W) A \ .$$

The conversion factors, f_D, are taken from (Rittmann 1983) whose values are based on internal dose conversion factors given in (DOE 1988).

Two decay times (5000 y and 10,000 y) were selected to give two consequence measures for the drinking water results. Reduction factors from retardation at these times are based on the contaminant flux kernel given in Appendix B. The results are shown in Table 4.

Table 4. Relative Rankings for Drinking Water Consequences.

Nuclide	DW (5000 y)	DW (10,000 y)
Tc-99	1.00E+00	1.00E+00
Se-79	3.09E-01	2.99E-01
Zr-93	2.97E-01	3.03E-01
U-233	1.11E-01	2.23E-01
U-234	1.02E-01	1.04E-01
Nb-93m	9.36E-02	9.50E-02
U-238	7.93E-02	8.09E-02
I-129	7.09E-02	7.24E-02
Ac-227	3.57E-02	6.73E-02
Pa-231	2.70E-02	5.08E-02
Ra-226	1.83E-02	3.70E-02
Cm-245	1.66E-02	1.13E-02
U-236	1.08E-02	1.61E-02
U-235	5.07E-03	6.58E-03
Ni-59	1.42E-03	1.39E-03
Pd-107	5.67E-04	5.78E-04
C-14	3.54E-04	1.97E-04
Cm-246	2.62E-04	1.28E-04
Sm-147	1.65E-04	1.68E-04
Nb-94	2.16E-05	1.85E-05
U-232	6.75E-06	6.68E-06

For typical Hanford conditions (see Appendix B) the travel time is long, even for unretarded radionuclides, so that even mild retardation greatly reduces the ranking. Compared to the unretarded travel time, the retarded travel time, t_a, is increased by the retardation factor, R_f. For the parameters given in Appendix B, the unretarded travel time is about 6000 y, and R_f is given by

$$R_f = 1+14k_d \ .$$

Thus, even a k_d as small as one implies a travel time near 90,000 y.

Table 5 shows the results if retardation is neglected. Long-lived nuclides such as ^{239}Pu, ^{240}Pu, and ^{237}Np now move to the top of the list. Neptunium is of particular interest since it may only be weakly retarded.

Table 5. Relative Rankings for Drinking Water Consequences. (no retardation)

Nuclide	DW (5000 y)	DW (10,000 y)
Pu-239	1.00E+00	1.00E+00
Pu-240	1.41E-01	9.59E-02
Np-237	2.81E-02	3.22E-02
Sn-126	4.12E-03	4.58E-03
Tc-99	3.80E-03	4.30E-03
Th-229	1.25E-03	4.95E-03
Se-79	1.18E-03	1.29E-03
Zr-93	1.13E-03	1.30E-03
Am-243	1.13E-03	8.12E-04
Am-241	8.84E-04	3.24E-06
Pb-210	4.25E-04	9.74E-04
U-233	4.21E-04	9.59E-04
U-234	3.88E-04	4.45E-04
Nb-93m	3.56E-04	4.08E-04
U-238	3.01E-04	3.48E-04
I-129	2.70E-04	3.11E-04
Ac-227	1.36E-04	2.89E-04
Pa-231	1.03E-04	2.18E-04
Ra-226	6.95E-05	1.59E-04
Cm-245	6.29E-05	4.84E-05
Th-230	4.89E-05	9.40E-05
U-236	4.12E-05	6.93E-05
Pu-242	2.36E-05	2.70E-05
U-235	1.93E-05	2.83E-05
Cs-135	6.24E-06	7.17E-06
Ni-59	5.41E-06	5.98E-06
Pd-107	2.15E-06	2.48E-06
C-14	1.34E-06	8.47E-07

Another consideration is that the peak dose may occur for times much greater than 10,000 years. For the peak dose, retardation still gives a large reduction (see Appendix B), not, however, as large as the reductions one sees for a fixed point in time.

Many of the radionuclides in this table have been the focus of previous studies including ^{99}Tc, ^{129}I, and ^{237}Np. Selenium 79 has also been previously recognized but is usually not as highly ranked; a key issue is whether or not it can be assumed to be retarded. Other nuclides that have drawn little previous attention include the uranium daughter products, ^{231}Pa and ^{227}Ac. Although their immediate parent ^{231}Th (see Figure 2) is retarded, it is not an effective barrier because of its short 26 half-life. The expected retardation of ^{231}Pa and ^{227}Ac needs to be confirmed.

Another nuclide in the list that merits additional study is ^{233}U. Although it is not dominant here, the current ORIGEN2 model did not include a significant amount of ^{233}U generation from thoria irradiations. Because they are more difficult to calculate, two other uranium isotopes, ^{234}U and ^{236}U, also merit further investigation.

The ranking of ^{14}C is artificially low. The calculation of ^{14}C is uncertain, and the current ^{14}C value should be adjusted upward to be more consistent with estimates from previous studies that have examined this problem in more detail. However, it still would not be highly ranked, most likely because of the 99% reduction credited in pretreatment.

4.2 RETARDATION AND LONG-TERM RADIOACTIVE DECAY

In most cases, application of the waste reduction factors is independent of the production and decay calculations carried out in ORIGEN2. An exception, noted earlier, is for the reprocessing separations that occur at an early time.

Another exception is for the long-lived uranium daughters. These include, among others, ^{231}Pa and ^{227}Ac, which are daughters of ^{235}U noted earlier. An artifact of applying the soil reduction factors at long times after the uranium isotopes have partially decayed is to increase the rank the corresponding daughters. For this reason, the ORIGEN2 calculation was modified to explicitly include the uranium pretreatment reduction factor shown in Table 3. To retain the simplicity of the ORIGEN2 run to the extent possible, all other reductions were treated as discussed.

4.2 INTRUDER

As for the drinking water consequence, the intruder consequence, denoted by Q_I, is determined by multiplying the activity A, computed by ORIGEN2, by the corresponding waste tank reduction factor r_T, the pretreatment reduction factor, r_G, and by the specific intruder dose conversion factor, f_I:

$$Q_I = f_I r_T r_G A$$

For the intruder consequences, no credit is taken for retardation in the soil. The conversion factors, f_I, are taken from (Rittman 1983) whose values are based on internal dose conversion factors taken from (DOE 1988). The results are shown in Table 6 for both 100 y and 500 y decay times.

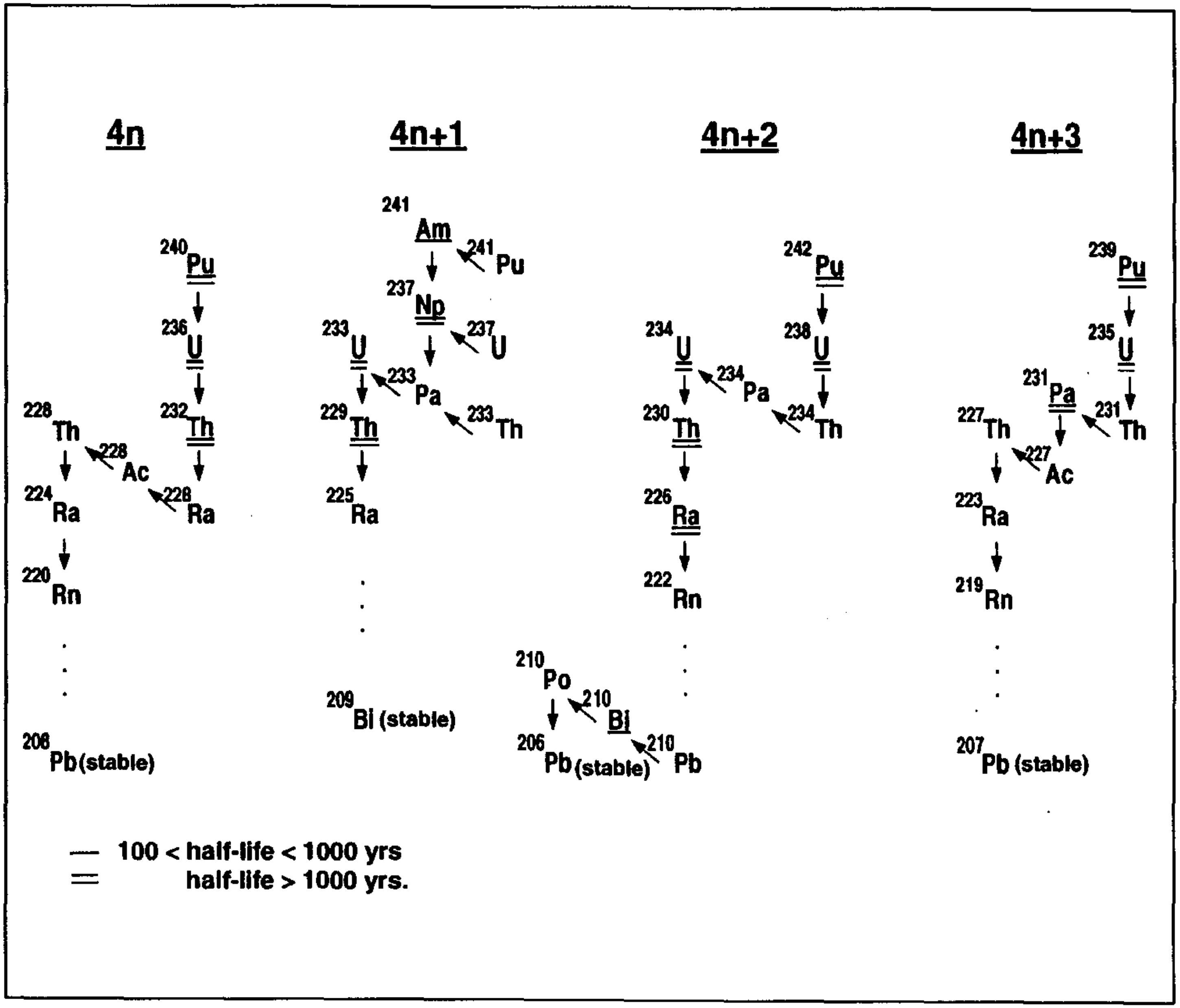


Figure 2. Actinide Decay Chain.

Table 6. Relative Rankings for Intruder Consequences.

Nuclide	Intruder (100 y)	Intruder (500 y)
Sr-90	1.00E+00	1.00E+00
Cs-137	7.00E-01	9.26E-01
Sn-126	1.35E-01	1.84E+03
Am-241	3.27E-02	2.36E+02
Pu-239	1.64E-02	2.22E+02
Tc-99	5.78E-03	0.79E+02
Pu-240	3.39E-03	4.43E+01
Np-237	9.59E-04	1.32E+01
Sm-151	8.67E-04	5.44E-01
Pu-238	5.72E-04	5.03E-01
Eu-154	1.81E-04	2.48E-14
Cm-244	1.35E-04	4.14E-07
Eu-152	1.17E-04	2.24E-09
Cd-113m	7.99E-05	6.07E-09
Am-242m	7.89E-05	1.74E-01
Am-243	4.05E-05	5.34E-01
Ni-63	2.52E-05	1.69E-02
Zr-93	2.45E-05	3.35E-01
U-238	2.39E-05	3.27E-01
Se-79	2.15E-05	2.92E-01
U-234	1.94E-05	2.79E-01
Nb-94	7.92E-06	1.07E-01
Cm-243	7.41E-06	6.02E-06
I-129	7.12E-06	9.73E-02
Nb-93m	7.10E-06	9.69E-02
Ac-227	2.34E-06	3.68E-02
Pu-241	2.08E-06	1.69E-05
U-235	1.81E-06	2.57E-02
Cm-245	1.59E-06	2.09E-02
Ra-226	1.00E-06	4.72E-02
C-14	8.95E-07	1.16E-02
Th-230	8.87E-07	1.42E-02
U-233	6.05E-07	3.17E-02
Ni-59	5.76E-07	7.83E-03

The results in Table 6 are sorted on the 100 y column. However, except for the short-lived nuclides, ^{90}Sr and ^{137}Cs, there is little difference between the 100 y and 500 y rankings for the top-ranked radionuclides. After strontium and cesium, ^{126}Sn is the dominant contributor followed by ^{241}Am and ^{239}Pu. These results are uncertain. The amount of ^{239}Pu is highly dependent upon on the assumed reprocessing efficiencies, and the ^{241}Am inventory includes all the N-Reactor production, some of which never enters the waste stream. Furthermore, in contrast to most radionuclides, N Reactor dominates the ^{241}Am production because of its relatively high burnup and long cooling times prior to reprocessing.

4.3 SHIELDING

A shielding consequence measure, Q_S, was obtained by taking the activities (decayed 20 y from 1990 to 2010) and multiplying them by the tank reduction factor, r_T, and a shielding consequence factor, f_S:

$$Q_S = f_S r_T A$$

The shielding consequence factor was defined as the relative dose that would penetrate a shield in plant operations and is given by

$$f_S = \sum_i a_i e^{-\frac{\mu(E_i)}{\rho}\rho x}$$

where a_i is the intensity of a photon of energy E_i, and the sum is over all photons for the given radionuclide. A generic form was used to approximate the mass attenuation coefficient:

$$\frac{\mu(E)}{\rho} = 0.04 e^{0.15[\ln(E)-\ln(8)]^2}$$

where the energy E is in MeV and μ/ρ has units cm^2/g.

To provide a gage of the sensitivity to the character of the shield, two shields were examined, a relatively thin shield of 0.635 cm (1/4") of iron and a thicker shield of 4.45 cm (1 3/4") of lead. (The values are sorted on the latter values.)

Table 7. Relative Rankings for Shielding Consequences.

Nuclide	1/4" Fe	1 3/4" Pb
Ba-137m (Cs-137)	0.257E+08	0.266E+06
Eu-154	0.539E+05	0.169E+04
Co-60	0.166E+04	0.812E+02
Eu-152	0.971E+03	0.282E+02
Sb-126m (Sn-126)	0.250E+04	0.224E+02
Kr-85	0.371E+04	0.141E+02
Sb-125	0.145E+04	0.686E+01
Sb-126 (Sb-126)	0.593E+03	0.665E+01
Cs-134	0.317E+03	0.425E+01
Ni-59	0.112E+03	0.101E+01
Y-90	0.629E+00	0.620E-01
Pm-146	0.376E+01	0.329E-01
Pa-233 (Np-237)	0.111E+03	0.262E-01
Np-238 (Am-242m)	0.542E+00	0.172E-01
Am-241	0.189E+02	0.473E-02

The rank of most nuclides is not especially sensitive to the shield type. Some, however, such as ^{241}Am with its relatively low-energy photons would be ranked higher for the thinner iron shield. Caution must be exercised in interpreting these results since they are based solely on the photons of the identified radionuclide. The top ranked entry ^{137m}Ba is, for example, the short-lived daughter of ^{137}Cs. (Parents of short-lived nuclides are noted in parenthesis in Table 7.) Also, while ^{90}Y (the short-lived daughter of ^{90}Sr) is also on the list, its ranking is underestimated because of neglected bremstrahhlung radiation.

Two europium isotopes show up prominently as does ^{60}Co. The latter is an activation product, difficult to estimate, and could be seriously in error. Note finally that ^{85}Kr is a noble gas.

5.0 CONCLUSIONS AND RECOMMENDATIONS

The motivation for the current effort was to provide a basis for ranking contaminants important to a low-level waste performance assessment. Starting from a new calculation of total radionuclide production, independent results were obtained with a clearly documented basis. Most previous studies have been based on TRAC or closely related work which focused more on transaction processing and individual tank inventories.

By focusing on global inventories, the current work should provide an excellent constraint on ongoing work that seeks to evaluate more detailed tank and process inventories. Thus, the methods and results are applicable to other phases of the cleanup program including high-level waste, plant design, and database activities.

In general, the results support previous studies of important radionuclides such as ^{99}Tc, ^{129}I, and others. Radionuclides such as ^{154}Eu important to plant design are also identified.

Several areas were found that need further consideration. A better estimate of ^{233}U production is required as well as for the minor uranium isotopes, ^{234}U and ^{236}U. Unless the retardation of ^{237}Np can be better established, further work is needed here, both in its production and its subsequent processing.

Based on the intruder scenario, the reduction, if any, of ^{126}Sn in the low-level waste stream needs to be established. The intruder results also place increased emphasis on the ^{239}Pu and ^{241}Am estimates.

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Appendix A ORIGEN2 Input Runstream

```
 92   1  0.995
 94   1  0.99
 92  10  0.06
-1
-1
-1
TIT     Composite Hanford Radionuclide Production
LIP     0 0 0
RDA     Assume MK IV Inner cross sections
RDA     Increase U235 Sigf to 100.0 barns
LPU     922350  -1
LIB     0 1 2 3 411 -412 413 9 3 0 1  0
RDA
RDA     1 Metric ton Natural uranium
INP     -1  1  -1  -1  1  1
RDA     1 Metric ton 0.95 % enriched uranium
INP     -2  1  -1  -1  1 1
RDA     Trace impurities for 1 metric ton on uranium
INP     -3  1  -1  -1  1 1
RDA     1 metric ton zircaloy
INP     -4  1  -1  -1  1 1
RDA     1 metric ton aluminum
INP     -5  1  -1  -1  1 1
RDA
RDA         << SINGLE PASS PRODUCTION >>
RDA     Create a vector with 1 metric ton of natural uranium
BAS     One metric ton of natural uranium
MOV     -1  1  0  1.0         URANIUM VECTOR
RDA     Add uranium trace impurities
ADD     -3  1  0  1.0
RDA     Add 5.0 wt% aluminum cladding
ADD     -5  1  0  0.05
RDA
RDA     Strategy is to irradiate a single MTU and then decay and
RDA     accumulate it to model the operating histogram.
RDA
RDA     Assume burnup of 800 MWd/MTU with a spec. power of 10 MW/MTU.
BUP
IRP     80.0    10.0    1 2 4 2
BUP
RDA     Decay for 100 days. Then extract 99% of Pu and  99.5% of U
RDA      (unrecovered to vector 5)
DEC     100.0            2 3 4 1
PRO     3  4  5    1
MOV     5  -7  0  1.0      Save results for single MTU
MOV     1  -6  0  1.0
RDA
RDA     Now begin decay and accumulation sequence
RDA     1st step includes 1944-1947.
MOV     -7  1  0  6.0              Norm. = 6 units for 1st step
DEC     4.0             1 2 5 1     Decay for 4 years
RDA
ADD     -7  2  0  16.0               1948-1951
DEC     4.0             2 3 5 1
ADD     -7  3  0  54.0               1952-1955
DEC     4.0             3 4 5 1
ADD     -7  4  0  148.0              1956-1959
DEC     4.0             4 5 5 1
ADD     -7  5  0  205.0              1960-1963
DEC     4.0             5 6 5 1
ADD     -7  6  0  180.0              1964-1967
DEC     4.0             6 7 5 1
ADD     -7  7  0  62.0               1968-1971
RDA     For the last step, assume the end of reprocessing takes place
RDA     (on an average) at the end of 1969.
RDA     Now decay 21 more years to take the result to end of 1990.
DEC     21.0            7 8 5 1
RDA
RDA     Save and normalize Single-pass results to a total 67.1E+6 MWd
MOV     8  -8  0  125.0        = 67.1E+6 / (671 x 800.0)
RDA
RDA    --------------- Preliminary output --------------
```

```
MOV     -8  3  0  1.0
MOV     -6  1  0  1.0
MOV     -7  2  0  1.0
CUT       5  1.0E-10       -1
OPTA     8 8 8 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTL     8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTF     8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OUT     3 1 -1  0
RDA     ---------------
STP     2
5  920000  1.0E+6           0  0.0
0
2  922350  0.00950E+6  922380  0.99002E+6    0 0.0
2  922340  0.00008E+6  922360  0.00040E+6    0 0.0
0
4 400000  19.0  290000  18.0     480000    0.19  40000    1.0
4 130000 790.0   10000   1.3     140000   93.0  280000   56.0
4 250000   9.2  120000   3.0     260000  330.0  240000   22.0
4  50000   0.14  70000   8.0      60000  520.0       0    0.0
0
4 130000  75.0  500000 1.45E+4   260000 1350.0  240000 1000.0
4 280000 550.0   60000 275.0     920000    2.5  400000 0.9814E+6
4  50000   0.5  480000   0.5     270000   10.0  290000   50.0
4 720000 200.0   10000  25.0     820000  100.0  120000   20.0
4 250000  50.0  420000  50.0      70000   80.0  140000  100.0
4 110000  20.0  220000  50.0     740000   50.0  230000   50.0
0
4 130000  1.0E+6      0   0.0
0
RDA     << N-REACTOR PRODUCTION >>
RDA
RDA     Assume MK IV Inner cross sections
RDA     Set U235 Sigf cross section to 55.0 barns
LPU     922350  -1
LIB     0 1 2 3 411 -412 413 9 4 0 1  0
RDA
RDA     Create a vector with 1 metric ton of 0.95% enriched uranium
BAS     One metric ton of 0.95% enriched uranium
MOV     -2  1  0  1.0      .  URANIUM VECTOR
RDA     Add uranium trace impurities
ADD     -3  1  0  1.0
RDA     Add 7.0 wt% zircaloy cladding
ADD     -4  1  0  0.07
RDA
RDA     Assume burnup of 2000 MWd/MTU with a spec. power of 10 MW/MTU.
BUP
IRP     100.0     10.0   1 2 4 2       100 day irrad.
DEC     120.0            2 3 4 0        20 day decay
IRP     220.0     10.0   3 4 4 0       100 day irrad.
BUP
RDA     Decay for 300 days. Then extract 99% of Pu and  99.5% of U
RDA      (unrecovered to vector 7)
DEC     300.0           4 5 4 1
PRO     5  6  7   1
MOV     7  -7  0  1.0     Save results for single MTU
MOV     1  -6  0  1.0
RDA
RDA     First step is for 1964-1967.
MOV     -7  1  0  22.0            22 units for 1st step
DEC     4.0            1 2 5 1    Decay for 4 years
RDA
ADD     -7  2  0  27.0            1968-1971
DEC     4.0            2 3 5 1
ADD     -7  3  0  29.0            1972-1975
DEC     4.0            3 4 5 1
ADD     -7  4  0  28.0            1976-1979
DEC     4.0            4 5 5 1
ADD     -7  5  0  21.0            1980-1983
DEC     4.0            5 6 5 1
ADD     -7  6  0  19.0            1984-1987
RDA     For the last step, assume the end of reprocessing takes place
RDA     (on an average) at the end of 1985.
RDA     Now decay 5 more years to take the result to end of 1990.
DEC     5.0            6 7 5 1
```

```
RDA
RDA     Save and normalize N Reactor results to a total 14.6E+6 MWd
MOV     7  -9  0  50.0      = 14.6E+6 / (146 x 2000.0)
RDA
RDA     Save N Reactor results at end of 1971 and normalize.
MOV     2   3  0  50.0
RDA     Decay 21 more years to take the result to end of 1990.
DEC     21.0      3 4 5 1
MOV     4  -10  0  1.0     save result
RDA
RDA    --------------- Preliminary output --------------
MOV     -9  3  0  1.0
MOV     -6  1  0  1.0
MOV     -7  2  0  1.0
CUT       5  1.0E-10      -1
OPTA     8 8 8 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTL     8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTF     8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OUT     3 1 -1  0
RDA    ---------------
STP     2
RDA     << Composite calculations >>
RDA
RDA     First, calculate Composite for end of 1971; store in vector 12
MOV     -8  12  0  1.0   Get single-pass values
ADD     -10 12  0  1.0   Add in N Reactor to end of 1971
HED     12   71cmpsit
RDA
RDA     Now do full composite calculations for all of N production
MOV     -8  1  0  1.0   Put single-pass values in vector 1
MOV     -9  2  0  1.0   Put N-reactor values in vector 2
MOV     1   3  0  1.0
ADD     2   3  0  1.0   Put composite values in vector 3
BAS     Total Hanford production (include pretreatment)
HED     1  Sngl-pass
HED     2  N-reactor
HED     3  Composite
RDA  Apply Pretreatment reduction factor for uranium
RDA  (Save result in vector 5; use 5 to start DK below)
PRO      3  5  6  10
RDA
RDA     Decay composite, assume end of 1990 is time t = 0.
RDA     Assumptions:
RDA       Operations in 2010 -                    +20
RDA       Closure in 2030                         +40
RDA       Intruder 1, Closure +100 years         +140
RDA       Intruder 2, Closure +500 years         +540
RDA       Closure +5000                         +5040
RDA       Closure +10,000                      +10040
RDA       Closure +20,000                      +20040
RDA       Closure +100,000                    +100040
RDA
DEC      20.0       5   4   5  1
DEC      40.0       4   5   5  0
DEC      140.0      5   6   5  0
DEC      540.0      6   7   5  0
DEC      5040.0     7   8   5  0
DEC      10040.0    8   9   5  0
DEC      20040.0    9  10   5  0
DEC      1.0E+5    10  11   5  0
RDA
CUT       5  1.0E-10    7  1.0E-20    -1
OPTA     8 8 8 8 7 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTL     8 8 8 8 7 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTF     8 8 8 8 7 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OUT     12  1 -1  0
STP   4
```

Appendix B Contaminant Retardation in Unsaturated Soils.

Simple formulas are developed to estimate the contaminant flux reduction factor that arises from retardation. The following concepts are presented in standard groundwater texts. Since the interpretation and use of these concepts varies somewhat, especially for unsaturated flow and transport, specific details are summarized here.

Contaminant flux in the vadose zone

The peak contaminant concentration in drinking water is proportional to the contaminant flux, $\Gamma_w(x,t)$, entering the aquifer from the vadose zone. The flux is calculated by convoluting the release rate R(t) with a flux kernel G(x,t):

$$\Gamma_w(x,t) = \int R(t')\, G(x, t-t')\, dt'$$

where the kernel is related to the infinite media Green's function, K(x,t), for the contaminant concentration by

$$G(x,t) = \left(\frac{x+u_e t}{2t}\right) K(x,t)$$

and where K(x,t) is given by

$$K(x,t) = \frac{1}{\sqrt{4\pi D_e t}}\, e^{-\frac{(x-u_e t)^2}{4D_e t}}$$

The pore velocity, u_e, and the dispersion coefficient, D_e, are effective values defined by

$$u_e = \frac{v}{\theta R_f},$$
$$D_e = \frac{D}{\theta R_f}$$

where v is the discharge velocity, and θ is the volumetric moisture content. The retardation factor R_f, in turn, is related to the distribution coefficient, k_d by

$$R_f = 1 + (\rho_B/\theta)\, k_d$$

where ρ_B is the bulk density.

There are two situations where the convolution integral can be approximated to give a simple result for the contaminant flux, Γ_w, when the release time is short compared to the spread in the kernel and when it is long. There are two distinct time scales for the flux kernel.

Diffusive and advective time scales are defined by the following relations:

$$t_d = \frac{x^2}{2D_e},$$
$$t_a = \frac{x}{u_e}$$

respectively. With these definitions, the flux kernel can be rewritten as

$$G(x,t) = \frac{1}{2}\left[\frac{1}{t} + \frac{1}{t_a}\right]\sqrt{\frac{t_d}{2\pi t}}\, e^{-\left(\frac{t_d}{t_a}\right)\frac{(t-t_a)^2}{2tt_a}}$$

Approximate peak flux for longer release times

For a slow release where the width of the kernel is relatively narrow in time, the kernel is approximately a delta function, and the flux is represented by the source release rate R(t) slightly dispersed. For a normalized source constant over a release period, T, the peak flux is simply $\Gamma_w = 1/T$. It is worth noting that this is only true when the character of the kernel is advective. For a diffusion-dominated kernel, the functional dependence at longer times (but still short compared to the advective time scale) is not controlled by the exponential decrease that is characteristic of the advective kernel.

Approximate peak flux for shorter release times

For a fast release where the source release rate function, R(t), is narrow, the kernel G(x,t), approximates the contaminant flux. For advective behavior, the peak in G(x,t) is near the peak in the exponential (i.e.,, for $t \approx t_a$). By inspection, the peak value for this condition is

$$\Gamma_w^{peak}(x,t) = \frac{1}{4t_a}\sqrt{\frac{t_d}{2\pi t_a}}$$

The purpose of this development is to obtain a simple expression for the impact of the distribution coefficient k_d. Unretarded time scales for diffusion and advection, $t_d(0)$ and $t_a(0)$, can be defined by factoring out the retardation factor R_f:

$$t_d = R_f t_d(0)\ ,$$
$$t_a = R_f t_a(0)$$

Note that the ratio t_d/t_a is thus independent of k_d. Since t_a is proportional to R_f the peak flux is thus inversely proportional to R_f

Numerical values

Typical parameter values will provide the explicit numerical values needed for this consequence ranking study. A volumetric moisture content θ = 0.1 mL/cm3, a recharge rate of v = 0.1 cm/y, and a vadose transport distance, x = 60 m are assumed leading to an advective transport time of

$$t_a(0) = 6000\ y\ .$$

Assume further that the effective dispersion coefficient, defined by factoring $1/R_f$ from D_e is given by $D_e(0) = 30\ cm^2/y$. This value yields a diffusive transport time large compared to the advective time:

$$t_d(0) = 600{,}000\ y\ .$$

These results lead to the following simple expression for the peak contaminant flux:

$$\Gamma_w^{peak}(k_d) = \frac{1}{R_f}\frac{1}{6000\ y}\ .$$

For this set of assumed numerical values, the unretarded contaminant flux is roughly equivalent to a unit source released over a 6000 y period. Retardation reduces the peak contaminant flux by the factor $1/R_f$. For the assumed volumetric moisture content of 0.1 and a bulk density of 1.4 g/cm^3, R_f becomes $(1+14k_d)$. Values for k_d near one and higher give significant reductions.

Appendix C Complete ORIGEN2 Curie Inventories

ACTIVATION, CURIES

	Sngl-pass	N-reactor	Composite	20.0YR	40.0YR	140.0YR	540.0YR	5040.0YR	1.0E+04YR	2.0E+04YR	1.0E+05YR	71cmpsit
NI 63	1.454e+04	6.884e+03	2.143e+04	1.843e+04	1.585e+04	7.462e+03	3.665e+02	6.914e-13	3.016e-29	0.000e+00	0.000e+00	1.673e+04
C 14	6.007e+02	2.618e+02	8.625e+02	8.604e+02	8.584e+02	8.480e+02	8.080e+02	4.688e+02	2.560e+02	7.635e+01	4.802e-03	6.885e+02
NI 59	1.450e+02	6.181e+01	2.068e+02	2.067e+02	2.067e+02	2.065e+02	2.058e+02	1.979e+02	1.895e+02	1.738e+02	8.694e+01	1.657e+02
CO 60	2.833e+01	1.606e+04	1.608e+04	1.159e+03	8.345e+01	1.618e-04	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.533e+03
SN121M	0.000e+00	1.331e+02	1.331e+02	1.009e+02	7.644e+01	1.910e+01	7.436e-02	0.000e+00	0.000e+00	0.000e+00	0.000e+00	4.022e+01
ZR 93	3.159e-02	2.923e+01	2.926e+01	2.926e+01	2.926e+01	2.926e+01	2.925e+01	2.919e+01	2.913e+01	2.900e+01	2.796e+01	9.841e+00
H 3	1.024e+02	1.553e+02	2.577e+02	8.385e+01	2.729e+01	9.957e-02	1.767e-11	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.350e+02
NB 93M	2.314e-02	1.519e+01	1.522e+01	2.326e+01	2.616e+01	2.779e+01	2.779e+01	2.773e+01	2.767e+01	2.755e+01	2.657e+01	6.575e+00
SB125	0.000e+00	1.644e+04	1.644e+04	1.103e+02	7.394e-01	1.002e-11	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	3.231e+02
FE 55	1.480e+03	2.636e+04	2.784e+04	1.346e+02	6.510e-01	1.720e-12	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.890e+03
MO 93	0.000e+00	2.116e-01	2.116e-01	2.108e-01	2.099e-01	2.058e-01	1.901e-01	7.796e-02	2.895e-02	3.991e-03	5.256e-10	7.092e-02
TE125M	0.000e+00	4.012e+03	4.012e+03	2.690e+01	1.804e-01	2.444e-12	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	7.884e+01
SR 90	1.539e-05	4.514e-02	4.516e-02	2.806e-02	1.743e-02	1.613e-03	1.182e-07	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.261e-02
Y 90	1.540e-05	4.516e-02	4.517e-02	2.806e-02	1.743e-02	1.613e-03	1.183e-07	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.261e-02
BE 10	9.534e-03	2.510e-03	1.204e-02	1.204e-02	1.204e-02	1.204e-02	1.204e-02	1.202e-02	1.199e-02	1.194e-02	1.153e-02	1.038e-02
TC 99	6.293e-10	9.766e-03	9.766e-03	9.765e-03	9.765e-03	9.762e-03	9.749e-03	9.607e-03	9.452e-03	9.149e-03	7.053e-03	3.278e-03
NB 94	7.329e-13	3.420e-04	3.420e-04	3.417e-04	3.415e-04	3.403e-04	3.357e-04	2.879e-04	2.427e-04	1.725e-04	1.125e-05	1.147e-04
AG108M	9.631e-05	9.214e-05	1.885e-04	1.690e-04	1.515e-04	8.777e-05	9.892e-06	2.135e-16	3.008e-28	0.000e+00	0.000e+00	1.260e-04
TA182	0.000e+00	4.450e-03	4.450e-03	1.182e-04	1.182e-04	1.182e-04	1.182e-04	1.181e-04	1.181e-04	1.180e-04	1.173e-04	3.967e-05
HF182	0.000e+00	1.182e-04	1.182e-04	1.182e-04	1.182e-04	1.182e-04	1.182e-04	1.181e-04	1.181e-04	1.180e-04	1.173e-04	3.967e-05
AG108	8.572e-06	8.200e-06	1.677e-05	1.504e-05	1.348e-05	7.812e-06	8.804e-07	1.900e-17	2.676e-29	0.000e+00	0.000e+00	1.121e-05
RE187	0.000e+00	1.295e-05	1.295e-05	1.295e-05	1.295e-05	1.295e-05	1.295e-05	1.295e-05	1.295e-05	1.295e-05	1.295e-05	4.345e-06
PB205	0.000e+00	1.218e-05	1.218e-05	1.218e-05	1.218e-05	1.218e-05	1.218e-05	1.218e-05	1.217e-05	1.217e-05	1.215e-05	4.087e-06
SI 32	4.993e-06	1.647e-06	6.640e-06	6.500e-06	6.363e-06	5.719e-06	3.733e-06	3.076e-08	1.487e-10	3.476e-15	0.000e+00	5.542e-06
P 32	4.994e-06	1.647e-06	6.641e-06	6.500e-06	6.363e-06	5.720e-06	3.734e-06	3.077e-08	1.487e-10	3.476e-15	0.000e+00	5.542e-06
LU176	0.000e+00	6.038e-09	6.038e-09	6.038e-09	6.038e-09	6.038e-09	6.038e-09	6.038e-09	6.038e-09	6.038e-09	6.038e-09	2.026e-09
TC 98	4.466e-18	4.638e-10	4.638e-10	4.638e-10	4.638e-10	4.638e-10	4.637e-10	4.634e-10	4.630e-10	4.622e-10	4.562e-10	1.556e-10
IR192	0.000e+00	7.465e-11	7.465e-11	7.043e-11	6.649e-11	4.987e-11	1.578e-11	3.779e-17	2.147e-23	0.000e+00	0.000e+00	2.451e-11
IR192M	0.000e+00	7.454e-11	7.454e-11	7.037e-11	6.644e-11	4.983e-11	1.577e-11	3.775e-17	2.146e-23	0.000e+00	0.000e+00	2.449e-11
AG109M	1.045e-04	1.455e-01	1.456e-01	2.654e-06	4.837e-11	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.363e-04
CD109	1.045e-04	1.455e-01	1.456e-01	2.654e-06	4.837e-11	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.363e-04
PT193	0.000e+00	2.357e-11	2.357e-11	2.292e-11	2.230e-11	1.941e-11	1.115e-11	2.179e-14	2.127e-17	2.029e-23	0.000e+00	7.830e-12
BI208	0.000e+00	1.576e-11	1.576e-11	1.576e-11	1.576e-11	1.576e-11	1.574e-11	1.561e-11	1.546e-11	1.518e-11	1.305e-11	5.289e-12
V 50	1.125e-13	1.107e-11	1.118e-11	1.118e-11	1.118e-11	1.118e-11	1.118e-11	1.118e-11	1.118e-11	1.118e-11	1.118e-11	3.827e-12
BI210M	0.000e+00	1.037e-11	1.037e-11	1.037e-11	1.037e-11	1.037e-11	1.037e-11	1.036e-11	1.035e-11	1.033e-11	1.014e-11	3.482e-12
TL206	0.000e+00	1.033e-11	1.033e-11	1.033e-11	1.033e-11	1.033e-11	1.033e-11	1.032e-11	1.031e-11	1.028e-11	1.010e-11	3.468e-12
PB204	0.000e+00	8.816e-12	8.816e-12	8.816e-12	8.816e-12	8.816e-12	8.816e-12	8.816e-12	8.816e-12	8.816e-12	8.816e-12	2.959e-12
TE123	0.000e+00	3.557e-12	3.557e-12	3.557e-12	3.557e-12	3.557e-12	3.557e-12	3.557e-12	3.557e-12	3.557e-12	3.557e-12	1.194e-12
IN115	2.227e-12	6.515e-13	2.878e-12	2.878e-12	2.878e-12	2.878e-12	2.878e-12	2.878e-12	2.878e-12	2.878e-12	2.878e-12	2.446e-12
MN 54	7.197e-04	7.311e+01	7.311e+01	6.716e-06	6.169e-13	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	9.603e-04
PD107	1.104e-13	9.865e-14	2.091e-13	2.091e-13	2.091e-13	2.091e-13	2.091e-13	2.090e-13	2.088e-13	2.086e-13	2.068e-13	1.435e-13
PO210	0.000e+00	2.604e-10	2.604e-10	4.149e-14	4.149e-14	4.149e-14	4.149e-14	4.145e-14	4.140e-14	4.130e-14	4.055e-14	1.393e-14
K 42	0.000e+00	2.051e-15	2.051e-15	1.347e-15	8.852e-16	1.083e-16	2.432e-20	0.000e+00	0.000e+00	0.000e+00	0.000e+00	5.852e-16
AR 42	0.000e+00	2.051e-15	2.051e-15	1.347e-15	8.851e-16	1.083e-16	2.432e-20	0.000e+00	0.000e+00	0.000e+00	0.000e+00	5.852e-16
SN119M	2.489e-16	6.263e+02	6.263e+02	6.638e-07	7.035e-16	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	5.852e-05
I129	0.000e+00	3.015e-16	3.015e-16	3.015e-16	3.015e-16	3.015e-16	3.015e-16	3.014e-16	3.013e-16	3.012e-16	3.002e-16	1.012e-16
AR 39	0.000e+00	1.402e-16	1.402e-16	1.332e-16	1.265e-16	9.775e-17	3.487e-17	3.211e-22	8.153e-28	0.000e+00	0.000e+00	4.617e-17

IR194	0.000e+00	6.813e-15	6.813e-15	6.759e-16	6.706e-17	6.446e-22	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	7.688e-16
OS194	0.000e+00	6.810e-15	6.810e-15	6.756e-16	6.704e-17	6.444e-22	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	7.685e-16
RH102	0.000e+00	6.125e-13	6.125e-13	5.140e-15	4.314e-17	1.797e-27	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.410e-14
ZN 65	1.961e-09	1.986e-02	1.986e-02	1.908e-11	1.833e-20	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	3.676e-09
AG110M	3.107e-13	5.585e-06	5.585e-06	8.848e-15	1.402e-23	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.030e-12
AG110	4.132e-15	7.428e-08	7.428e-08	1.177e-16	1.864e-25	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.370e-14
CA 45	2.929e-24	3.304e-05	3.304e-05	1.059e-18	2.282e-32	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	7.520e-16
SUMTOT	1.690e+04	7.112e+04	8.802e+04	2.117e+04	1.716e+04	8.593e+03	1.438e+03	7.237e+02	5.024e+02	3.067e+02	1.415e+02	2.160e+04

ACTINIDES, CURIES

	Sngl-pass	N-reactor	Composite	20.0YR	40.0YR	140.0YR	540.0YR	5040.0YR	1.0E+04YR	2.0E+04YR	1.0E+05YR	71cmpsit
AM241	1.567e+04	5.411e+04	6.978e+04	7.183e+04	7.118e+04	6.152e+04	3.240e+04	2.389e+01	7.597e-02	3.012e-02	4.433e-05	3.419e+04
PU239	3.272e+04	6.617e+03	3.934e+04	3.931e+04	3.929e+04	3.918e+04	3.873e+04	3.403e+04	2.947e+04	2.209e+04	2.208e+03	3.494e+04
PU241	5.290e+04	1.571e+05	2.100e+05	8.019e+04	3.062e+04	2.487e+02	1.478e-01	1.024e-01	6.810e-02	3.012e-02	4.433e-05	8.838e+04
PU240	5.256e+03	2.916e+03	8.172e+03	8.164e+03	8.151e+03	8.069e+03	7.734e+03	4.800e+03	2.825e+03	9.783e+02	2.034e-01	6.235e+03
PU238	1.483e+03	2.803e+03	4.286e+03	3.691e+03	3.180e+03	1.523e+03	9.802e+01	5.568e-08	6.980e-18	0.000e+00	0.000e+00	2.374e+03
CM244	1.120e+02	6.374e+03	6.486e+03	3.017e+03	1.403e+03	3.054e+01	6.852e-06	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.684e+03
PA233	3.587e+02	1.514e+02	5.101e+02	5.106e+02	5.110e+02	5.132e+02	5.190e+02	5.248e+02	5.239e+02	5.222e+02	5.089e+02	4.096e+02
NP237	3.587e+02	1.514e+02	5.101e+02	5.106e+02	5.110e+02	5.132e+02	5.190e+02	5.248e+02	5.239e+02	5.222e+02	5.089e+02	4.096e+02
AM242M	2.065e+01	2.507e+02	2.714e+02	2.477e+02	2.261e+02	1.433e+02	2.313e+01	2.835e-08	3.554e-18	0.000e+00	0.000e+00	1.020e+02
AM242	2.055e+01	2.495e+02	2.700e+02	2.465e+02	2.250e+02	1.426e+02	2.301e+01	2.821e-08	3.536e-18	0.000e+00	0.000e+00	1.015e+02
CM242	1.699e+01	2.174e+02	2.343e+02	2.038e+02	1.861e+02	1.179e+02	1.903e+01	2.340e-08	2.933e-18	0.000e+00	0.000e+00	8.393e+01
AM243	4.168e+00	4.472e+01	4.888e+01	4.879e+01	4.870e+01	4.825e+01	4.647e+01	3.045e+01	1.904e+01	7.443e+00	4.076e-03	1.917e+01
NP239	4.168e+00	4.472e+01	4.888e+01	4.879e+01	4.870e+01	4.825e+01	4.647e+01	3.045e+01	1.904e+01	7.443e+00	4.076e-03	1.917e+01
U234	1.365e+02	1.732e+01	1.539e+02	9.457e+00	9.651e+00	1.029e+01	1.086e+01	1.088e+01	1.086e+01	1.081e+01	1.047e+01	1.424e+02
CM243	4.425e-01	2.400e+01	2.444e+01	1.503e+01	9.238e+00	8.116e-01	4.835e-05	0.000e+00	0.000e+00	0.000e+00	0.000e+00	7.107e+00
PA234M	1.399e+02	1.213e+01	1.521e+02	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	1.440e+02
U238	1.399e+02	1.213e+01	1.521e+02	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	1.440e+02
TH234	1.399e+02	1.213e+01	1.521e+02	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	9.123e+00	1.440e+02
NP238	1.033e-01	1.254e+00	1.357e+00	1.239e+00	1.131e+00	7.166e-01	1.157e-01	1.418e-10	1.777e-20	0.000e+00	0.000e+00	5.100e-01
PU242	1.435e-01	6.903e-01	8.339e-01	8.354e-01	8.369e-01	8.423e-01	8.498e-01	8.446e-01	8.371e-01	8.222e-01	7.125e-01	3.754e-01
U237	1.298e+00	3.854e+00	5.152e+00	1.967e+00	7.511e-01	6.100e-03	3.621e-06	2.508e-06	1.668e-06	7.381e-07	1.086e-09	2.168e+00
TH231	5.724e+00	6.072e-01	6.331e+00	3.806e-01	3.814e-01	3.853e-01	4.006e-01	5.616e-01	7.177e-01	9.698e-01	1.650e+00	5.928e+00
U235	5.724e+00	6.072e-01	6.331e+00	3.806e-01	3.814e-01	3.853e-01	4.006e-01	5.616e-01	7.177e-01	9.698e-01	1.650e+00	5.928e+00
U236	2.534e+00	1.744e+00	4.278e+00	2.615e-01	2.663e-01	2.904e-01	3.839e-01	1.203e+00	1.754e+00	2.269e+00	2.536e+00	3.120e+00
TH230	1.567e-01	4.633e-02	2.030e-01	2.046e-01	2.063e-01	2.151e-01	2.528e-01	6.753e-01	1.124e+00	1.960e+00	6.400e+00	1.726e-01
CM245	1.518e-03	1.527e-01	1.542e-01	1.539e-01	1.537e-01	1.524e-01	1.475e-01	1.022e-01	6.798e-02	3.007e-02	4.415e-05	5.272e-02
U233	4.583e-02	1.028e-02	5.611e-02	4.800e-02	9.267e-02	3.165e-01	1.218e+00	1.143e+01	2.252e+01	4.394e+01	1.830e+02	5.091e-02
PA231	1.726e-02	5.361e-03	2.262e-02	2.277e-02	2.292e-02	2.368e-02	2.680e-02	6.832e-02	1.260e-01	2.648e-01	1.280e+00	1.909e-02
RA223	9.532e-03	2.071e-03	1.160e-02	1.684e-02	1.969e-02	2.332e-02	2.681e-02	6.834e-02	1.260e-01	2.649e-01	1.280e+00	1.047e-02
RN219	9.532e-03	2.071e-03	1.160e-02	1.684e-02	1.969e-02	2.332e-02	2.681e-02	6.834e-02	1.260e-01	2.649e-01	1.280e+00	1.047e-02
PO215	9.532e-03	2.071e-03	1.160e-02	1.684e-02	1.969e-02	2.332e-02	2.681e-02	6.834e-02	1.260e-01	2.649e-01	1.280e+00	1.047e-02
BI211	9.532e-03	2.071e-03	1.160e-02	1.684e-02	1.969e-02	2.332e-02	2.681e-02	6.834e-02	1.260e-01	2.649e-01	1.280e+00	1.047e-02
PB211	9.532e-03	2.071e-03	1.160e-02	1.684e-02	1.969e-02	2.332e-02	2.681e-02	6.834e-02	1.260e-01	2.649e-01	1.280e+00	1.047e-02
AC227	9.528e-03	2.068e-03	1.160e-02	1.683e-02	1.967e-02	2.332e-02	2.681e-02	6.834e-02	1.260e-01	2.649e-01	1.280e+00	1.047e-02
TL207	9.505e-03	2.065e-03	1.157e-02	1.679e-02	1.963e-02	2.326e-02	2.674e-02	6.815e-02	1.257e-01	2.642e-01	1.277e+00	1.044e-02
TH227	9.401e-03	2.043e-03	1.144e-02	1.661e-02	1.941e-02	2.300e-02	2.644e-02	6.740e-02	1.243e-01	2.613e-01	1.262e+00	1.033e-02
PA234	1.819e-01	1.576e-02	1.977e-01	1.186e-02	1.186e-02	1.186e-02	1.186e-02	1.186e-02	1.186e-02	1.186e-02	1.186e-02	1.872e-01
RA226	1.760e-03	3.161e-04	2.077e-03	3.817e-03	5.557e-03	1.426e-02	4.928e-02	4.617e-01	9.176e-01	1.772e+00	6.421e+00	1.916e-03
RN222	1.760e-03	3.161e-04	2.077e-03	3.817e-03	5.557e-03	1.426e-02	4.928e-02	4.617e-01	9.176e-01	1.772e+00	6.421e+00	1.916e-03
PO218	1.760e-03	3.161e-04	2.077e-03	3.817e-03	5.557e-03	1.426e-02	4.928e-02	4.617e-01	9.176e-01	1.772e+00	6.421e+00	1.916e-03
BI214	1.760e-03	3.160e-04	2.076e-03	3.816e-03	5.556e-03	1.425e-02	4.927e-02	4.616e-01	9.174e-01	1.772e+00	6.419e+00	1.915e-03
PB214	1.760e-03	3.160e-04	2.076e-03	3.816e-03	5.556e-03	1.425e-02	4.927e-02	4.616e-01	9.174e-01	1.772e+00	6.419e+00	1.915e-03
PO214	1.760e-03	3.160e-04	2.076e-03	3.815e-03	5.554e-03	1.425e-02	4.926e-02	4.615e-01	9.173e-01	1.771e+00	6.418e+00	1.915e-03
CM246	1.082e-05	3.367e-03	3.377e-03	3.367e-03	3.358e-03	3.309e-03	3.120e-03	1.614e-03	7.758e-04	1.792e-04	1.465e-09	1.139e-03
PB210	5.938e-04	7.480e-05	6.686e-04	1.764e-03	3.157e-03	1.147e-02	4.926e-02	4.615e-01	9.173e-01	1.771e+00	6.418e+00	6.386e-04
BI210	5.938e-04	7.482e-05	6.686e-04	1.764e-03	3.157e-03	1.147e-02	4.926e-02	4.615e-01	9.173e-01	1.771e+00	6.418e+00	6.386e-04
PO210	5.938e-04	7.482e-05	6.686e-04	1.764e-03	3.157e-03	1.147e-02	4.926e-02	4.615e-01	9.173e-01	1.771e+00	6.418e+00	6.386e-04
TH228	1.960e-02	2.720e-02	4.680e-02	3.130e-03	2.616e-03	1.089e-03	1.671e-04	1.446e-04	1.407e-04	1.335e-04	9.307e-05	2.634e-02
RA224	1.960e-02	2.726e-02	4.686e-02	3.130e-03	2.616e-03	1.089e-03	1.671e-04	1.446e-04	1.407e-04	1.335e-04	9.307e-05	2.634e-02

RN220	1.960e-02	2.726e-02	4.686e-02	3.130e-03	2.616e-03	1.089e-03	1.671e-04	1.446e-04	1.407e-04	1.335e-04	9.307e-05	2.634e-02
BI212	1.960e-02	2.726e-02	4.686e-02	3.130e-03	2.616e-03	1.089e-03	1.671e-04	1.446e-04	1.407e-04	1.335e-04	9.307e-05	2.634e-02
PO216	1.960e-02	2.726e-02	4.686e-02	3.130e-03	2.616e-03	1.089e-03	1.671e-04	1.446e-04	1.407e-04	1.335e-04	9.307e-05	2.634e-02
PB212	1.960e-02	2.726e-02	4.686e-02	3.130e-03	2.616e-03	1.089e-03	1.671e-04	1.446e-04	1.407e-04	1.335e-04	9.307e-05	2.634e-02
U232	1.915e-02	2.003e-02	3.919e-02	3.043e-03	2.545e-03	1.062e-03	1.665e-04	1.444e-04	1.401e-04	1.319e-04	8.144e-05	2.578e-02
PO212	1.256e-02	1.746e-02	3.002e-02	2.005e-03	1.676e-03	6.976e-04	1.071e-04	9.264e-05	9.013e-05	8.553e-05	5.963e-05	1.688e-02
NP236	7.658e-04	8.876e-04	1.653e-03	1.653e-03	1.653e-03	1.652e-03	1.648e-03	1.604e-03	1.556e-03	1.465e-03	9.049e-04	1.064e-03
TL208	7.043e-03	9.793e-03	1.684e-02	1.125e-03	9.398e-04	3.912e-04	6.005e-05	5.195e-05	5.054e-05	4.796e-05	3.344e-05	9.465e-03
FR223	1.315e-04	2.854e-05	1.600e-04	2.322e-04	2.714e-04	3.218e-04	3.700e-04	9.431e-04	1.739e-03	3.656e-03	1.767e-02	1.445e-04
RA225	6.776e-05	1.180e-05	7.955e-05	1.281e-04	2.608e-04	2.184e-03	3.064e-02	2.340e+00	8.020e+00	2.449e+01	1.845e+02	7.428e-05
PB209	6.776e-05	1.180e-05	7.955e-05	1.281e-04	2.608e-04	2.184e-03	3.064e-02	2.340e+00	8.020e+00	2.449e+01	1.845e+02	7.428e-05
AT217	6.776e-05	1.180e-05	7.955e-05	1.281e-04	2.608e-04	2.184e-03	3.064e-02	2.340e+00	8.020e+00	2.449e+01	1.845e+02	7.428e-05
FR221	6.776e-05	1.180e-05	7.955e-05	1.281e-04	2.608e-04	2.184e-03	3.064e-02	2.340e+00	8.020e+00	2.449e+01	1.845e+02	7.428e-05
TH229	6.776e-05	1.179e-05	7.955e-05	1.281e-04	2.608e-04	2.184e-03	3.064e-02	2.340e+00	8.020e+00	2.449e+01	1.845e+02	7.428e-05
AC225	6.776e-05	1.180e-05	7.955e-05	1.281e-04	2.608e-04	2.184e-03	3.064e-02	2.340e+00	8.020e+00	2.449e+01	1.845e+02	7.428e-05
BI213	6.776e-05	1.180e-05	7.955e-05	1.281e-04	2.608e-04	2.184e-03	3.064e-02	2.340e+00	8.020e+00	2.449e+01	1.845e+02	7.428e-05
PO213	6.630e-05	1.154e-05	7.784e-05	1.253e-04	2.552e-04	2.137e-03	2.998e-02	2.290e+00	7.847e+00	2.396e+01	1.805e+02	7.268e-05
PU236	7.169e-04	3.143e-02	3.215e-02	3.962e-04	1.507e-04	1.487e-04	1.483e-04	1.444e-04	1.401e-04	1.319e-04	8.144e-05	1.425e-03
PO211	2.669e-05	5.799e-06	3.249e-05	4.716e-05	5.512e-05	6.530e-05	7.507e-05	1.914e-04	3.529e-04	7.418e-04	3.584e-03	2.933e-05
TL209	1.464e-06	2.548e-07	1.718e-06	2.767e-06	5.634e-06	4.717e-05	6.618e-04	5.055e-02	1.732e-01	5.290e-01	3.985e+00	1.605e-06
TH232	1.326e-08	2.357e-08	3.684e-08	3.709e-08	3.735e-08	3.873e-08	4.538e-08	2.288e-07	5.995e-07	1.614e-06	1.163e-05	2.139e-08
AC228	1.160e-08	1.774e-08	2.934e-08	3.604e-08	3.711e-08	3.873e-08	4.538e-08	2.288e-07	5.995e-07	1.614e-06	1.163e-05	1.880e-08
RA228	1.160e-08	1.773e-08	2.933e-08	3.604e-08	3.711e-08	3.873e-08	4.538e-08	2.288e-07	5.995e-07	1.614e-06	1.163e-05	1.880e-08
PU244	5.248e-10	1.382e-08	1.435e-08	1.435e-08	1.435e-08	1.435e-08	1.435e-08	1.435e-08	1.435e-08	1.435e-08	1.434e-08	5.164e-09
U240	5.241e-10	1.381e-08	1.433e-08	1.433e-08	1.433e-08	1.433e-08	1.433e-08	1.433e-08	1.433e-08	1.433e-08	1.432e-08	5.158e-09
NP240M	5.241e-10	1.381e-08	1.433e-08	1.433e-08	1.433e-08	1.433e-08	1.433e-08	1.433e-08	1.433e-08	1.433e-08	1.432e-08	5.158e-09
CM247	1.152e-12	1.084e-09	1.085e-09	1.085e-09	1.085e-09	1.085e-09	1.085e-09	1.085e-09	1.085e-09	1.084e-09	1.080e-09	3.650e-10
PU243	1.152e-12	1.084e-09	1.085e-09	1.085e-09	1.085e-09	1.085e-09	1.085e-09	1.085e-09	1.085e-09	1.084e-09	1.080e-09	3.650e-10
CF249	5.295e-14	4.446e-10	4.447e-10	4.280e-10	4.114e-10	3.376e-10	1.530e-10	2.087e-14	1.059e-18	2.727e-27	0.000e+00	1.473e-10
CM248	9.471e-14	2.702e-10	2.703e-10	2.703e-10	2.703e-10	2.703e-10	2.700e-10	2.676e-10	2.649e-10	2.595e-10	2.204e-10	9.079e-11
CF250	6.447e-15	2.830e-10	2.830e-10	9.807e-11	3.398e-11	1.698e-13	1.853e-19	1.548e-19	1.268e-19	8.516e-20	3.523e-21	6.112e-11
CF251	1.228e-17	8.738e-13	8.738e-13	8.604e-13	8.472e-13	7.843e-13	5.760e-13	1.786e-14	3.766e-16	1.674e-19	0.000e+00	2.916e-13
NP235	1.205e-07	3.269e-03	3.269e-03	9.168e-09	2.571e-14	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	2.836e-07
CF252	0.000e+00	8.591e-13	8.591e-13	4.485e-15	2.342e-17	9.088e-29	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.413e-14
BK249	1.130e-19	2.413e-10	2.413e-10	3.244e-17	4.362e-24	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.076e-15
AM245	1.639e-24	3.500e-15	3.500e-15	4.706e-22	6.327e-29	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.561e-20
SUMTOT	1.095e+05	2.311e+05	3.406e+05	2.081e+05	1.556e+05	1.121e+05	8.020e+04	4.004e+04	3.352e+04	2.444e+04	5.005e+03	1.695e+05

FISSION PRODUCTS, CURIES

	Sngl-pass	N-reactor	Composite	20.0YR	40.0YR	140.0YR	540.0YR	5040.0YR	1.0E+04YR	2.0E+04YR	1.0E+05YR	71cmpsit
CS137	1.095e+08	3.241e+07	1.419e+08	8.937e+07	5.630e+07	5.585e+06	5.410e+02	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.185e+08
BA137M	1.035e+08	3.066e+07	1.342e+08	8.454e+07	5.326e+07	5.284e+06	5.118e+02	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.121e+08
Y 90	9.587e+07	2.568e+07	1.216e+08	7.551e+07	4.691e+07	4.341e+06	3.182e+02	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.030e+08
SR 90	9.585e+07	2.568e+07	1.215e+08	7.550e+07	4.690e+07	4.340e+06	3.182e+02	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.030e+08
SM151	3.315e+06	5.678e+05	3.882e+06	3.328e+06	2.853e+06	1.321e+06	6.065e+04	5.374e-11	1.012e-27	0.000e+00	0.000e+00	3.494e+06
KR 85	3.915e+06	1.873e+06	5.788e+06	1.588e+06	4.358e+05	6.748e+02	3.952e-09	0.000e+00	0.000e+00	0.000e+00	0.000e+00	4.277e+06
TC 99	2.976e+04	6.430e+03	3.619e+04	3.619e+04	3.619e+04	3.618e+04	3.613e+04	3.561e+04	3.503e+04	3.391e+04	2.614e+04	3.192e+04
H 3	2.111e+05	1.055e+05	3.165e+05	1.030e+05	3.352e+04	1.223e+02	2.170e-08	0.000e+00	0.000e+00	0.000e+00	0.000e+00	2.332e+05
EU154	1.039e+05	2.190e+05	3.229e+05	6.441e+04	1.285e+04	4.061e+00	4.050e-14	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.400e+05
ZR 93	4.267e+03	8.709e+02	5.138e+03	5.138e+03	5.138e+03	5.137e+03	5.137e+03	5.126e+03	5.114e+03	5.091e+03	4.910e+03	4.559e+03
NB 93M	3.126e+03	4.530e+02	3.579e+03	4.411e+03	4.711e+03	4.880e+03	4.880e+03	4.870e+03	4.859e+03	4.837e+03	4.665e+03	3.321e+03
CD113M	1.986e+04	1.023e+04	3.009e+04	1.163e+04	4.498e+03	3.887e+01	2.168e-07	0.000e+00	0.000e+00	0.000e+00	0.000e+00	2.218e+04
SN126	1.348e+03	3.499e+02	1.698e+03	1.698e+03	1.698e+03	1.696e+03	1.692e+03	1.640e+03	1.584e+03	1.478e+03	8.491e+02	1.466e+03
SB126M	1.348e+03	3.499e+02	1.698e+03	1.698e+03	1.698e+03	1.696e+03	1.692e+03	1.640e+03	1.584e+03	1.478e+03	8.491e+02	1.466e+03
CS135	9.603e+02	1.887e+02	1.149e+03	1.149e+03	1.149e+03	1.149e+03	1.149e+03	1.147e+03	1.146e+03	1.142e+03	1.115e+03	1.024e+03
SE 79	8.992e+02	1.935e+02	1.093e+03	1.093e+03	1.092e+03	1.091e+03	1.086e+03	1.036e+03	9.817e+02	8.824e+02	3.759e+02	9.642e+02
EU155	1.187e+05	1.387e+05	2.574e+05	1.572e+04	9.605e+02	8.172e-04	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.305e+05
EU152	1.321e+03	2.059e+03	3.381e+03	1.220e+03	4.402e+02	2.692e+00	3.773e-09	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.774e+03
SB126	1.887e+02	4.899e+01	2.377e+02	2.377e+02	2.377e+02	2.375e+02	2.368e+02	2.296e+02	2.217e+02	2.069e+02	1.189e+02	2.052e+02
PM147	7.257e+05	7.203e+06	7.928e+06	4.022e+04	2.040e+02	6.841e-10	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	8.416e+05
SN121M	1.541e+02	5.821e+01	2.124e+02	1.609e+02	1.219e+02	3.046e+01	1.186e-01	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.717e+02
PD107	8.073e+01	3.225e+01	1.130e+02	1.130e+02	1.130e+02	1.130e+02	1.130e+02	1.129e+02	1.129e+02	1.127e+02	1.118e+02	9.156e+01
I129	5.686e+01	1.393e+01	7.079e+01	7.079e+01	7.079e+01	7.079e+01	7.079e+01	7.077e+01	7.076e+01	7.073e+01	7.048e+01	6.154e+01
SB125	4.022e+04	4.456e+05	4.858e+05	3.257e+03	2.184e+01	2.959e-10	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	4.898e+04
TE125M	9.817e+03	1.087e+05	1.185e+05	7.948e+02	5.329e+00	7.221e-11	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.195e+04
CS134	3.083e+03	3.447e+05	3.478e+05	4.184e+02	5.076e-01	1.277e-15	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	5.008e+03
PM146	1.317e+01	5.389e+01	6.706e+01	5.393e+00	4.337e-01	1.458e-06	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.856e+01
C 14	2.396e-01	5.205e-02	2.916e-01	2.909e-01	2.902e-01	2.867e-01	2.732e-01	1.585e-01	8.656e-02	2.582e-02	1.624e-06	2.570e-01
NB 94	9.521e-02	4.418e-02	1.394e-01	1.393e-01	1.392e-01	1.387e-01	1.369e-01	1.174e-01	9.894e-02	7.032e-02	4.585e-03	1.100e-01
HO166M	6.167e-02	6.086e-02	1.225e-01	1.211e-01	1.197e-01	1.130e-01	8.970e-02	6.667e-03	3.713e-04	1.150e-06	1.006e-26	8.201e-02
CE142	5.786e-02	1.226e-02	7.012e-02	7.012e-02	7.012e-02	7.012e-02	7.012e-02	7.012e-02	7.012e-02	7.012e-02	7.012e-02	6.197e-02
RB 87	5.403e-02	1.063e-02	6.466e-02	6.466e-02	6.466e-02	6.466e-02	6.466e-02	6.466e-02	6.466e-02	6.466e-02	6.466e-02	5.760e-02
SM147	2.141e-02	3.918e-03	2.533e-02	2.552e-02	2.552e-02	2.552e-02	2.552e-02	2.552e-02	2.552e-02	2.552e-02	2.552e-02	2.278e-02
AG108M	9.555e-03	6.787e-03	1.634e-02	1.465e-02	1.314e-02	7.612e-03	8.579e-04	1.851e-14	2.609e-26	0.000e+00	0.000e+00	1.174e-02
BE 10	5.961e-03	1.293e-03	7.254e-03	7.254e-03	7.254e-03	7.253e-03	7.252e-03	7.238e-03	7.222e-03	7.191e-03	6.946e-03	6.395e-03
EU150	2.057e-03	2.327e-03	4.384e-03	2.983e-03	2.029e-03	2.959e-04	1.338e-07	0.000e+00	0.000e+00	0.000e+00	0.000e+00	2.730e-03
AG108	8.504e-04	6.041e-04	1.455e-03	1.304e-03	1.169e-03	6.775e-04	7.635e-05	1.648e-15	2.322e-27	0.000e+00	0.000e+00	1.045e-03
TC 98	4.611e-04	2.943e-04	7.554e-04	7.554e-04	7.554e-04	7.554e-04	7.553e-04	7.548e-04	7.542e-04	7.529e-04	7.430e-04	5.599e-04
RH102	7.841e-02	3.338e+00	3.416e+00	2.868e-02	2.407e-04	1.003e-14	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.552e-01
KR 81	1.210e-04	7.875e-05	1.998e-04	1.997e-04	1.997e-04	1.997e-04	1.994e-04	1.965e-04	1.932e-04	1.870e-04	1.436e-04	1.474e-04
SM146	1.419e-05	1.054e-05	2.472e-05	2.652e-05	2.666e-05	2.667e-05	2.667e-05	2.667e-05	2.667e-05	2.667e-05	2.665e-05	1.809e-05
ND144	2.667e-06	5.724e-07	3.239e-06	3.239e-06	3.239e-06	3.239e-06	3.239e-06	3.239e-06	3.239e-06	3.239e-06	3.239e-06	2.859e-06
RH106	4.083e+01	6.848e+05	6.848e+05	7.290e-01	7.761e-07	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	5.667e+01
RU106	4.083e+01	6.848e+05	6.848e+05	7.290e-01	7.761e-07	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	5.667e+01
LA138	3.905e-07	7.263e-08	4.631e-07	4.631e-07	4.631e-07	4.631e-07	4.631e-07	4.631e-07	4.631e-07	4.631e-07	4.631e-07	4.149e-07
IN115	1.067e-07	2.283e-08	1.295e-07	1.295e-07	1.295e-07	1.295e-07	1.295e-07	1.295e-07	1.295e-07	1.295e-07	1.295e-07	1.144e-07
SM149	1.832e-08	1.603e-09	1.993e-08	1.993e-08	1.993e-08	1.993e-08	1.993e-08	1.993e-08	1.993e-08	1.993e-08	1.993e-08	1.886e-08
SM148	1.159e-08	7.024e-09	1.861e-08	1.861e-08	1.861e-08	1.861e-08	1.861e-08	1.861e-08	1.861e-08	1.861e-08	1.861e-08	1.394e-08
GD152	3.688e-10	2.547e-10	6.235e-10	6.993e-10	7.267e-10	7.420e-10	7.421e-10	7.421e-10	7.421e-10	7.421e-10	7.421e-10	4.626e-10

PR144	3.810e+00	8.541e+05	8.541e+05	1.569e-02	2.881e-10	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	4.589e+00
CE144	3.810e+00	8.540e+05	8.540e+05	1.569e-02	2.881e-10	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	4.588e+00
TE123	1.553e-11	3.193e-11	4.746e-11	4.746e-11	4.746e-11	4.746e-11	4.746e-11	4.746e-11	4.746e-11	4.746e-11	4.746e-11	2.624e-11
TM171	8.537e-10	7.304e-06	7.305e-06	5.344e-09	3.909e-12	8.190e-28	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	2.889e-08
PR144M	4.572e-02	1.025e+04	1.025e+04	1.882e-04	3.457e-12	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	5.506e-02
CD109	3.449e-09	5.372e-05	5.372e-05	9.793e-10	1.785e-14	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.521e-08
AG109M	3.449e-09	5.372e-05	5.372e-05	9.793e-10	1.785e-14	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.521e-08
AG110M	1.116e-05	1.001e+02	1.001e+02	1.586e-07	2.513e-16	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	2.407e-05
SN119M	2.689e-05	6.711e+01	6.711e+01	7.112e-08	7.538e-17	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	3.316e-05
AG110	1.485e-07	1.332e+00	1.332e+00	2.110e-09	3.343e-18	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	3.201e-07
GD153	3.361e-08	1.126e+00	1.126e+00	9.229e-10	7.565e-19	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	0.000e+00	1.193e-07
SUMTOT	4.132e+08	1.286e+08	5.418e+08	3.301e+08	2.068e+08	2.092e+07	1.145e+05	5.148e+04	5.071e+04	4.921e+04	3.921e+04	4.460e+08